SWISRUS PROJECT

NOVOVORONEZSH UNIT 5

LEVEL-2/3 PSA

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Introduction to Phase 2 of SWISRUS Project

- The work presented is being performed in the framework of Phase 2 of SWISRUS Project initiated in 1997
- NVNPP-5 PSA is being performed in accordance with the agreement between Gosatomnadzor of Russia and Federal Nuclear Safety Inspectorate of Switzerland

Assistance in Training in Level-2/3 PSA Methods

- Energy Research Inc. was providing support in:
  - Training in performance of Level-2 PSA
  - Reviewing results of the work done
  - Consulting in development of particular tasks of Level-2 PSA
  - Assistance in providing software and documentation requirements
- NVNPP specialists provided collection of information required for analyses

Main Tasks of Level-2/3 PSA

Tasks of Level-2 PSA：

- Studying processes of severe accidents progression
- Analysis of containment integrity subject to various loads (mechanical, thermal, etc.) occurring during severe accidents
- Identification of “weak points” of containment during severe accidents
- Identification of main modes of containment failure and estimation of radiological releases
- Development of information basis for Accident Management Strategies applicable to mitigation of severe accidents and radiological releases

Main Tasks of Level-2/3 PSA (cont.)

Tasks of Level-2 PSA：

- **Development of data/information for accident management procedures and measures that can potentially lead to reduction of risk**

Tasks of Level-3 PSA：

- **Analysis of propagation, dispersion and deposition of radionuclides**
- **Assessment of radioactivity doses received by population**
- **Analysis of efficiency of accident mitigation measures on the radioactivity doses received by population in case of severe accident**
- **Quantitative assessment of radioactive releases impact to health effects**

Interface Between Level-1 and Level-2 PSAs

MCSs (from Level-1 PSA) were grouped into Plant Damage States (PDS) from the point of view of similar impact on containment

- **General criteria for definition of PDSs are:**
  - **Accident progression prior to core damage**
  - **Active safety systems important for containment**

- **NVNPP-5 PDS characteristics were developed.**

- **End states of accident sequences were allocated to particular PDSs (in accordance with PDS characteristics). As a result, a PDS matrix was developed.**

Severe Accidents Analysis and Estimation of Containment Integrity

1. For dominating PDSs, the analysis of severe accident progression was performed using the MELCOR code

2. Accident Progression Event Trees (APETs) were developed for Level-2 PSA, which characterize severe accident progression and containment failure modes

3. Development of APETs was based on consideration of all PDSs, various phenomena during severe accidents that influence containment integrity and quantity of radiological releases

4. Special software is used for APET modeling quantification considering various dependencies in phenomena

Severe Accidents Analysis and Estimation of Containment Integrity (Cont.)

5. The APET quantification is based on
   - Level-1 PSA results
   - Information on PDS
   - Expert judgement
   - Design features of the plant
   - Analysis of specific phenomena occurring during severe accidents and their influence on containment integrity (for instance, hydrogen combustion)
   - Dependencies between different questions in APETs

Severe Accidents Analysis and Estimation of Containment Integrity (Cont.)

6. The following physical phenomena were considered:
   - Combustion of hydrogen in the containment
   - In-vessel and Ex-vessel steam explosion
   - High pressure ejection of core debris into containment compartments and direct containment heating
   - Basemat attack
   - Rocketing effect of reactor vessel in case of vessel bottom failure at high pressure
   - Creep-rupture of the primary system due to high temperatures (at high pressure)

Severe Accidents Analysis and Estimation of Containment Integrity (Cont.)

7. The results of severe accident progression are a large number of end states characterizing containment states.

End states were grouped into 9 categories (called "release category") on the basis of similarity in characteristics (7 characteristics were used) of radiological releases.

8. On the basis of results obtained, the likelihood of containment failure is estimated

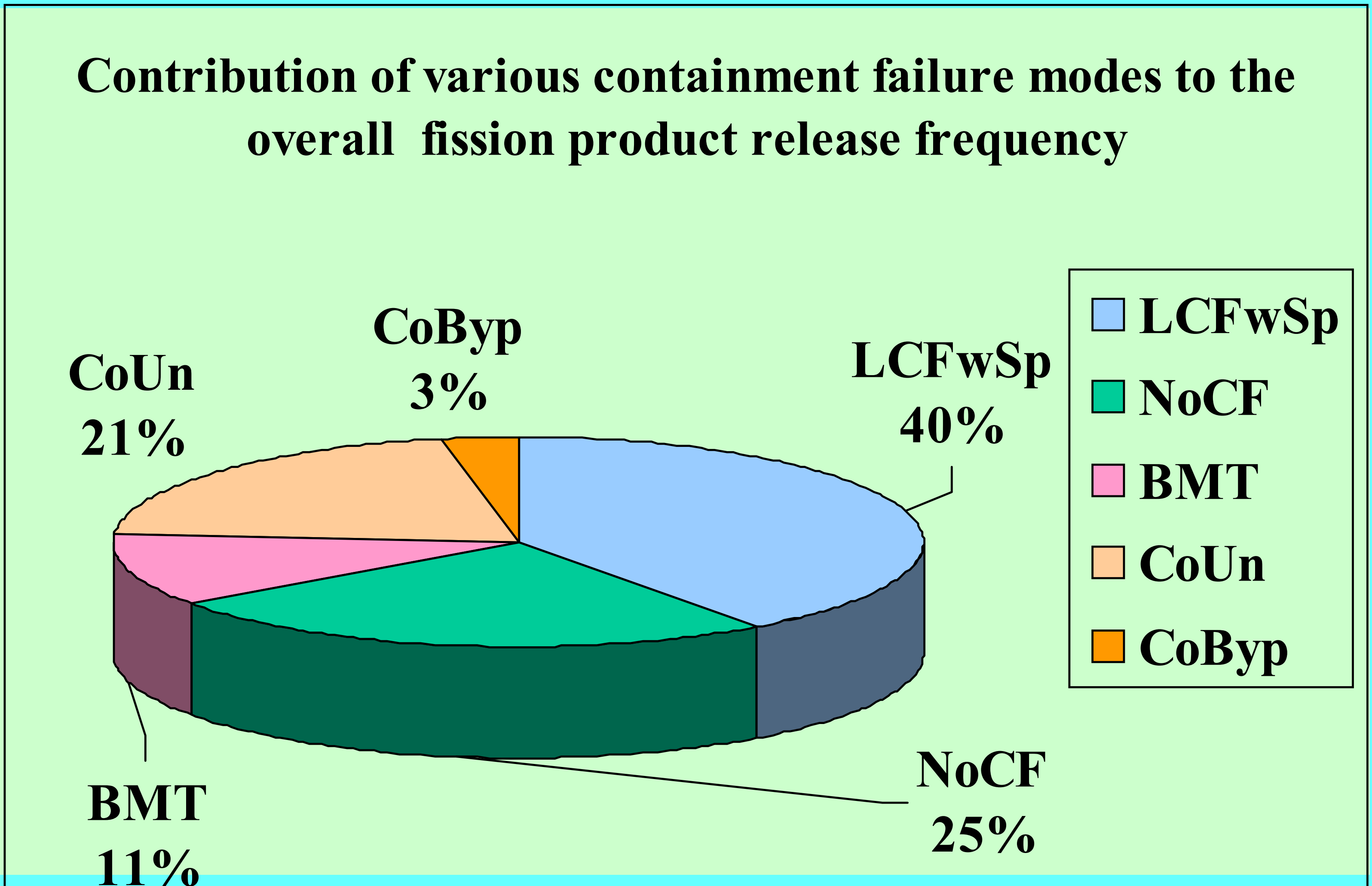
Note: That all the results presented herein, are based on:

- A fragility curve with the 5% at 4.6 bar & 95% at 9 bars)

Severe Accidents Analysis and Estimation of Containment Integrity (Cont.)

Containment Failure Mode	CDF, 1/reactor * year	% Contribution
Early containment failure with SS operation (ECFwSp)	1.1E-6	0
Early containment failure without SS (ECFnoSp)	0	0
Late containment failure with SS operation (LCFwSp)	**2.7E-4**	**40%**
Late containment failure without SS operation (LCFnoSp)	2E-7	0
No containment failure (NoCF)	**1.7E-4**	**25%**
Basement melt through (BMT)	**7.2E-5**	**11%**
Containment isolation system failure (CoUn)	**1.4E-4**	**21%**
High temperature induced SG tube/header failure	4E-8	0
Leak outside containment (CoByp)	**2.3E-5**	**3%**

Severe Accidents Analysis and Estimation of Containment Integrity (Cont.)



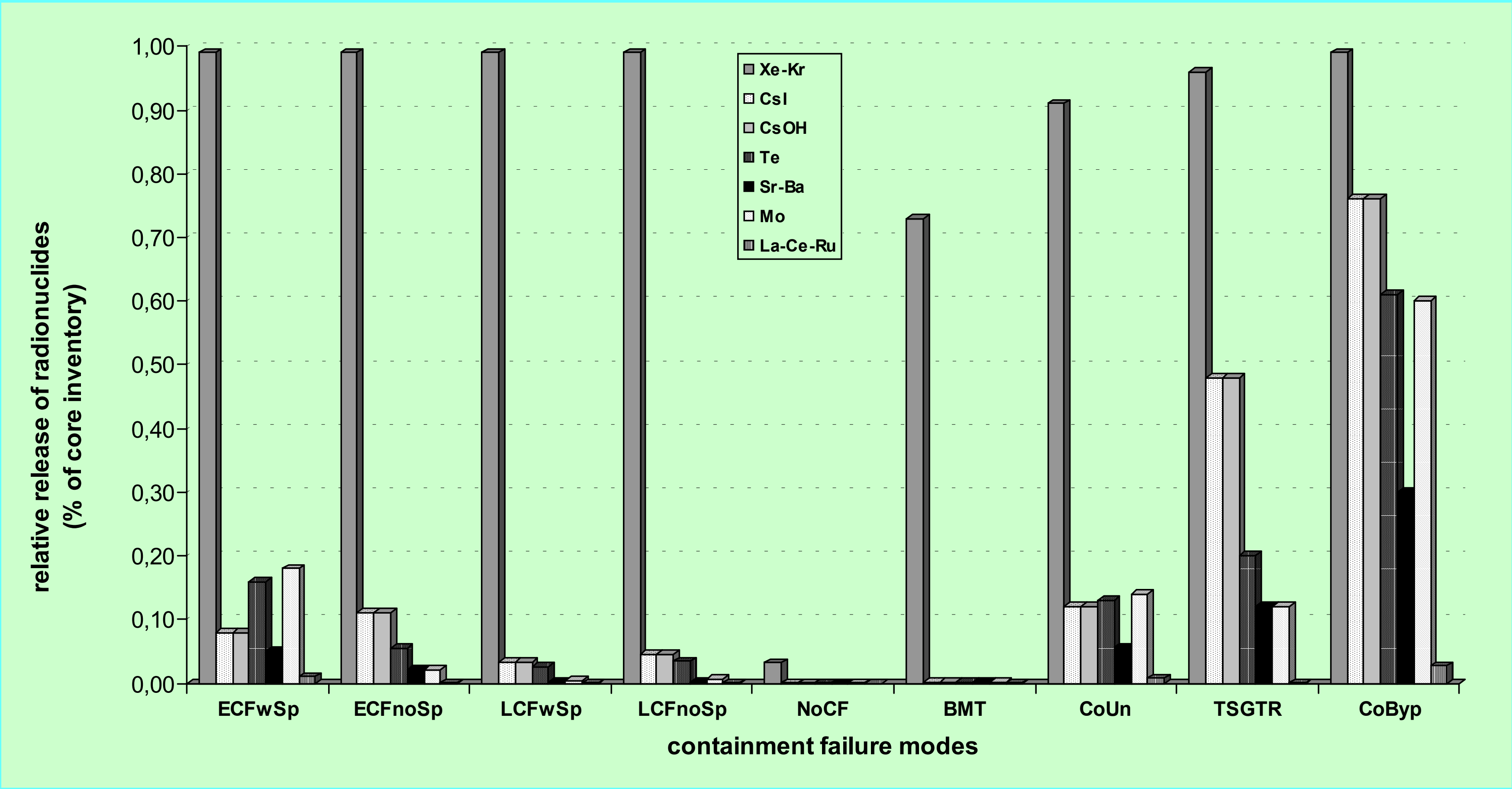
ACCIDENT SOURCE TERMS

1. The environmental release quantities (source term) associated with each release category (containment failure modes) were estimated .

2. In the present study, the MELCOR and ERPRA-ST(specially modified by Energy Research, Inc. for application to NVNPP-5) computer codes are used to analyze the radiological source terms.

3. Seven radiological groups were used to characterize the core radiological inventory and its release to the environment for NVNPP Unit 5.

ACCIDENT SOURCE TERMS



ACCIDENT SOURCE TERMS

4. The following releases are considered and were calculated In the present study:

   - In-vessel Releases,

     - Fission Product Transport in Reactor Coolant System,

     - Ex-Vessel Releases,

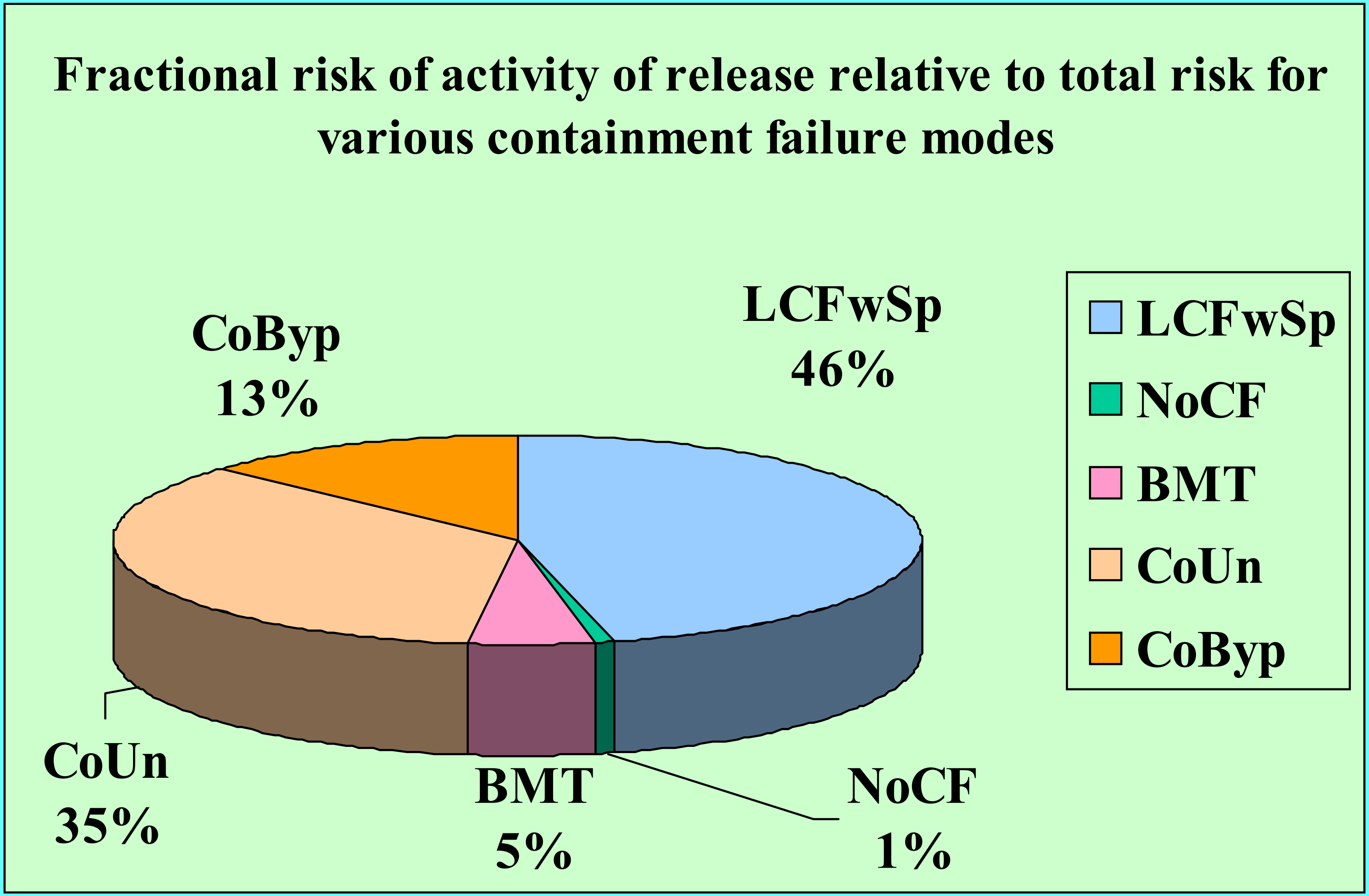
     - Fission Product Transport Inside Containment ,

     - Environmental Releases.

5. The risk associated with the activity of release of radionuclides to the environment, during severe accidents were calculated.

6. In the present study ERPRA computer code was used to calculate risk of activity.

ACCIDENT SOURCE TERMS



SENSITIVITY ANALYSIS

1. The following severe accident management actions were considered:

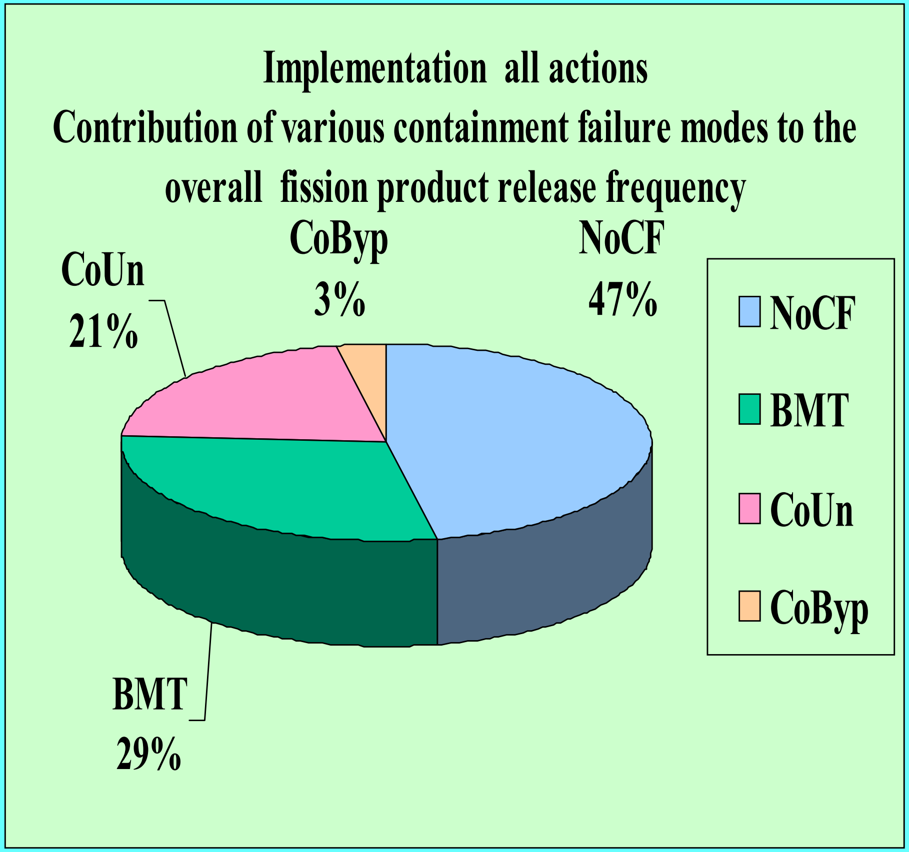
Action 1 - Addition of Water to Degrading Core to Prevent Vessel Breach

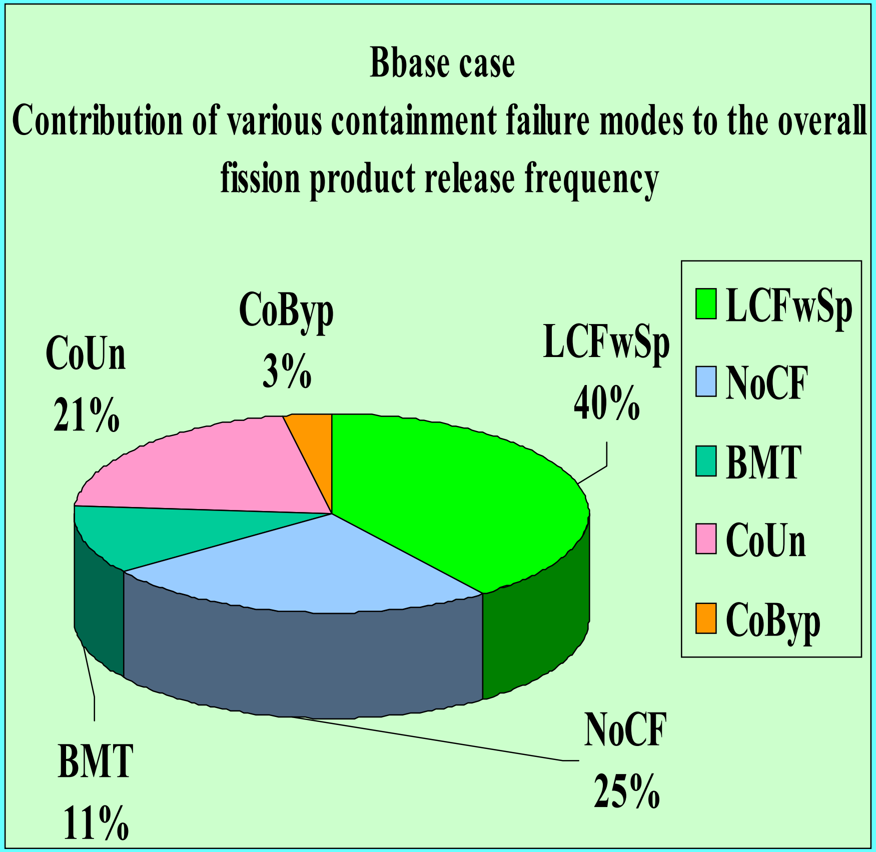
Action 2 - Addition of Water to Secondary Side of Steam Generators

Action 3 - Closure of Containment Isolation Valves

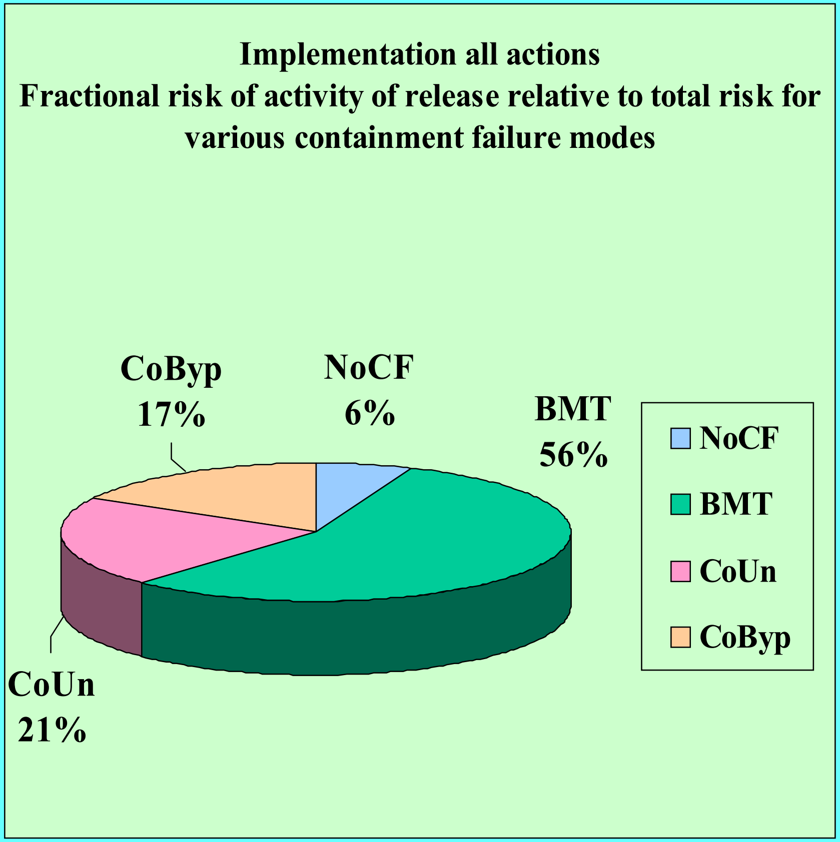
Action 4 - Hydrogen Control Via an Active System

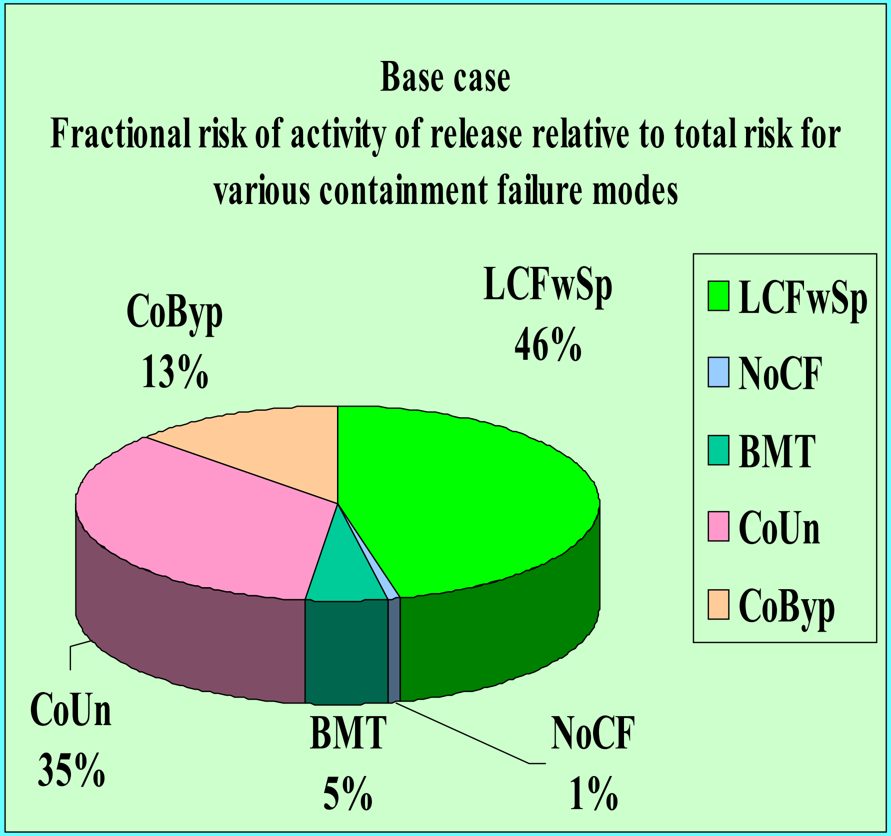
SENSITIVITY ANALYSIS





SENSITIVITY ANALYSIS





Interface Between Level-2 and Level-3 PSAs

1. The Level-2 PSA results have been used for analysis of off-site consequences of radioactive releases
2. The following characteristics of releases assessed in Level-2 PSA were used for the analysis of off-site consequences of radioactive releases
   - quantities of radionuclides,
   - composition of releases,
   - energy of releases,
   - height of releases
3. The analysis of off-site consequences of radioactive releases was performed for all containment failure modes assessed in Level-2 PSA

ANALYSIS OF OFF-SITE CONSEQUENCES OF RADIOACTIVE RELEASES

1. The MACCS computer code (Sandia National Laboratories) was used for the analysis of off-site consequences
2. The input deck was developed for MACCS on the basis of the following data:
   - Characteristics of releases
   - Meteorology data
   - Population distribution data
   - Accident mitigation measures
3. Main results of the analysis:
   - The propagation, dispersion and deposition of radionuclides in the environment has been considered

ANALYSIS OF OFF-SITE CONSEQUENCES OF RADIOACTIVE RELEASES (Cont.)

- ➤ The doses received by population are calculated as a function of distance from the plant
- ➤ The assessment of the impact of accident mitigation measures to the doses was performed
- ➤ The numbers of «early» and «late» fatalities were calculated for different containment failure modes and distances from the plant

4. For conducting the analysis, it was assumed that in the boundary of 30 km zone from the plant, 95% of population would be evacuated, and 5 % would remain in place

☛ Main calculation results are presented in Tables 1 and 2

SUMMARY

1. Main containment failure modes for NVNPP-5 were analyzed:
   - Late containment failure due to hydrogen and CO combustion
   - Containment bypass (leak from primary into secondary side)
   - Reactor basemat melt-through,
   - Containment isolation system failure
2. Risk of release of activity was calculated for major containment failure modes

SUMMARY (Cont.)

3. The most important containment failure modes, for which off-site consequences are most severe for NVNPP-5, are:
   - ➤ the leak from primary to secondary side with SGSV stuck open
   - ➤ containment isolation system failure
   - ➤ containment failure induced by hydrogen and carbon monoxide combustion
4. Risk sensitivity analyses were performed:
   - ➤ impact of implementation of severe accident management actions
   - ➤ impact of implementation of accident mitigation measures